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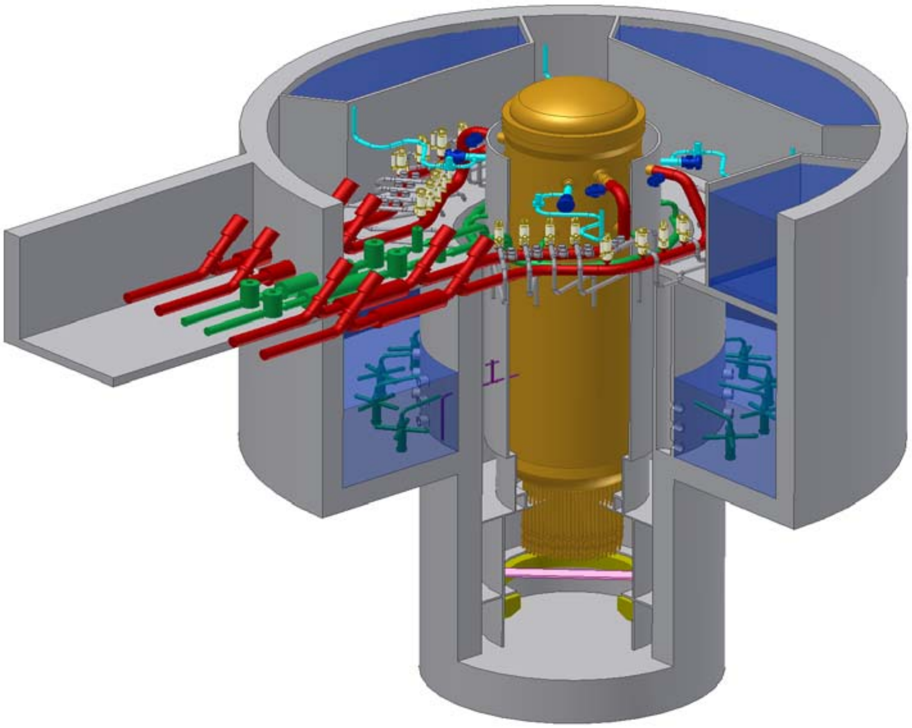

ESBWR Design Control Document

Tier 2

Chapter 1

Introduction and General Description of Plant

Appendices 1A-1D

Contents

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Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
A/D	Analog-to-Digital
AASHTO	American Association of Highway and Transportation Officials
AB	Auxiliary Boiler
ABS	Auxiliary Boiler System
ABWR	Advanced Boiling Water Reactor
ac / AC	Alternating Current
AC	Air Conditioning
ACF	Automatic Control Function
ACI	American Concrete Institute
ACS	Atmospheric Control System
AD	Administration Building
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFIP	Automated Fixed In-Core Probe
AGMA	American Gear Manufacturer's Association
AHS	Auxiliary Heat Sink
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
API	American Petroleum Institute
APRM	Average Power Range Monitor
APR	Automatic Power Regulator
APRS	Automatic Power Regulator System
ARI	Alternate Rod Insertion
ARMS	Area Radiation Monitoring System
ASA	American Standards Association
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ASTM	American Society of Testing Methods
AT	Unit Auxiliary Transformer

Global Abbreviations And Acronyms List

Term	Definition
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BOP	Balance of Plant
BPU	Bypass Unit
BPV	Bypass Valve
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative absolute velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBGAHVS	Control Building General Area
CBHVAC	Control Building HVAC
CBHVS	Control Building Heating, Ventilation and Air Conditioning System
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio

Global Abbreviations And Acronyms List

Term	Definition
CPS	Condensate Purification System
CPU	Central Processing Unit
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAHVS	Control Room Habitability Area HVAC Sub-system
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
DBE	Design Basis Event
dc / DC	Direct Current
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor
DG	Diesel-Generator
DHR	Decay Heat Removal
DM&C	Digital Measurement and Control
DOF	Degree of freedom
DOI	Dedicated Operators Interface
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DS	Independent Spent Fuel Storage Installation

Global Abbreviations And Acronyms List

Term	Definition
DTM	Digital Trip Module
DW	Drywell
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective full power years
EFU	Emergency Filter Unit
EHC	Electrohydraulic Control (Pressure Regulator)
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDDI	Fiber Distributed Data Interface
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive

Global Abbreviations And Acronyms List

Term	Definition
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPE	Fire Pump Enclosure
FTDC	Fault-Tolerant Digital Controller
FTS	Fuel Transfer System
FW	Feedwater
FWCS	Feedwater Control System
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GE-NE	GE Nuclear Energy
GEN	Main Generator System
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter
GM-B	Beta-Sensitive GM Detector
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HEP	Human error probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System

Global Abbreviations And Acronyms List

Term	Definition
HPT	High-pressure turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator
HWC	Hydrogen Water Chemistry
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IFTS	Inclined Fuel Transfer System
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program

Global Abbreviations And Acronyms List

Term	**Definition**
LAPP	Loss of Alternate Preferred Power
LCO	Limiting Conditions for Operation
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LD&IS	Leak Detection and Isolation System
LERF	Large early release frequency
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MS	Main Steam

Global Abbreviations And Acronyms List

Term	Definition
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NICWS	Nuclear Island Chilled Water Subsystem
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-service
ORNL	Oak Ridge National Laboratory

Global Abbreviations And Acronyms List

Term	Definition
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak cladding temperature
PCV	Primary Containment Vessel
PFD	Process Flow Diagram
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack
PSD	Power Spectra Density
PSS	Process Sampling System
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller

Global Abbreviations And Acronyms List

Term	Definition
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual heat removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square
RMS	Radiation Monitoring Subsystem
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWBCR	Radwaste Building Control Room
RWBGA	Radwaste Building General Area
RWBHVAC	Radwaste Building HVAC
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error

Global Abbreviations And Acronyms List

Term	Definition
RWM	Rod Worth Minimizer
SA	Severe Accident
SAR	Safety Analysis Report
SB	Service Building
S/C	Digital Gamma-Sensitive GM Detector
SC	Suppression Chamber
S/D	Scintillation Detector
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SMU	SSLC Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator

Global Abbreviations And Acronyms List

Term	Definition
SRP	Standard Review Plan
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Sum of the squares
SRV	Safety Relief Valve
SRVDL	Safety relief valve discharge line
SSAR	Standard Safety Analysis Report
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SSPC	Steel Structures Painting Council
ST	Spare Transformer
STI	Startup Test Instruction
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static water head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TEAS	Turbine Building Air Supply
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC
TBV	Turbine Bypass Valve
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine first stage pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-history accelerograph

Global Abbreviations And Acronyms List

Term	Definition
TLOS	Turbine Lubricating Oil System
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
TTWFATBV	Turbine trip with failure of all bypass valves
UBC	Uniform Building Code
UHS	Ultimate heat sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
USE	Upper Shelf Energy
USM	Uniform Support Motion
USMA	Uniform support motion response spectrum analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero period acceleration

APPENDIX 1A RESPONSE TO TMI RELATED MATTERS

Table 1A-1 addresses the TMI Action Plan Items listed in 10 CFR 50.34(f). Because the ESBWR includes design features different from the active plants considered in 10 CFR 50.34(f), consideration is given to all issues, in order to identify comparable ESBWR features which may address the issues.

1A.1 REFERENCES

1A-1 U. S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC report NUREG-0660, Vols. 1 and 2, May 1980.

1A-2 U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," USNRC Report NUREG-0737, November 1980.

1A-3 U. S. Nuclear Regulatory Commission, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," NUREG-0718, Revision 1, June 1981.

1A-4 Letter from D. B. Waters, Chairman, BWR Owners' Group, to D. G. Eisenhut, NRC, "BWR Owners' Group Evaluation of NUREG-0737 Requirements II.K.3.16 and II.K.3.18," March 31, 1981.

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
10 CFR 50. 34(f)(1)(i)	II.B.8	Levels 1 (Plant), 2 (Containment) & 3 (Site) PRAs to confirm meeting NRC Safety Goals.	A plant specific Probabilistic Risk Assessment (PRA) performed on the ESBWR design evaluates the plant in terms of core damage frequency and containment integrity. The PRA supports the design effort and establishes the capability of the design to meet established safety goals. The PRA contains Level 1 (Plant), Level 2 (Containment), and Level 3 (Site) PRA evaluations including internal and external events. In addition the PRA identified a number of design changes that improve the design of the ESBWR and which have been incorporated.	ESBWR PRA, Chapter 19
10 CFR 50. 34(f)(1)(ii)	II.E.1.1	PWR Auxiliary Feedwater System evaluation.	Applicable to PWRs only. The ESBWR does not have comparable systems.	N/A
10 CFR 50. 34(f)(1)(iii)	II.K.2.16 and II.K.3.25	Reactor Coolant Pump Seal damage.	ESBWR has no Reactor Coolant Pump; ESBWR is a passive plant and utilizes natural circulation to drive coolant flow.	N/A
10 CFR 50. 34(f)(1)(iv)	II.K.3.2	Power Operated Relief Valves	Not Applicable. ESBWR uses pneumatic externally actuated or spring-operated safety relief valves.	5.2.2, and 5.4.13
10 CFR 50. 34(f)(1)(v)	II.K.3.13	Separate HPCI/HPCS and RCIC system initiation levels such that RCIC initiates at a higher water level	The comparable ESBWR systems are the Automatic Depressurization System (ADS) / Gravity Driven Cooling System (GDCS) and	5.4.6, 5.4.7,

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		than HPCI/HPCS.	the Isolation Condenser System (ICS). The ICS initiates at a higher level (L2) than the ADS/GDCS (Level 1.5 or 1). High pressure inventory control and reactor decay heat removal following reactor isolation for the ESBWR is by means of the Isolation Condenser System (ICS). The ESBWR ICS replaces the traditional HPCI and RCIC Systems found in most BWRs, thus eliminating concerns about cold water injection and system initiation. The ICS initiates automatically on high reactor pressure or low reactor water level (Level 2). ICS also initiate on Loss of All Feedwater or on closure of the MSIVs whenever the reactor mode switch is in the RUN position. ESBWR low pressure inventory control is via the Gravity-Driven Cooling System (GDCS) in conjunction with the Automatic Depressurization System (ADS), which initiates at a lower water level (Level 1.5 or 1) than the ICS.	6.3.2.7, and 6.3.2.8.
10 CFR 50. 34(f)(1)(vi)	II.K.3.16	Perform a study to identify practical system modifications that would reduce challenges and failures of relief valves, without compromising	One of the key design criteria of the ESBWR is that SRVs should not need to open during any Anticipated Operational Occurrences (transients) or DBAs to protect	5.2.2.

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		the performance of the valves or other systems. (Applicable to BWR's only).	against overpressure. SRVs are only expected to open in the event of an ATWS or beyond design basis events. This is achieved through the use of the Isolation Condensers System (ICS). General Electric and the BWR Owners' Group responded to this requirement for earlier BWR models. Based on a review of the existing operating information on the challenge rate of relief valves, they concluded that the BWR/6 product line had already achieved the "order of magnitude" level of reduction in SRV challenge rate. The principal reason for this reduction is that the BWR/6 uses direct acting SRVs, not the pilot operated design used in some earlier BWRs. The ESBWR also uses direct acting SRVs.	
10 CFR 50. 34(f)(1)(vii)	II.K.3.18	Perform a feasibility and risk assessment study to determine the optimum Automatic Depressurization System (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling.	The ESBWR ADS does not require manual activation to ensure adequate core cooling. Actuation of ADS equipment is performed automatically upon receipt of a persistent reactor water Level 1.5 or 1 signal without need for operator action. Manual actuation is also possible. Automatic ADS complements manual ADS.	5.2.2.2, 6.3.2.8, and 7.3.1.1.

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			Subsection 7.3.1 describes the logic and sequencing of the ADS in detail. For the above reasons, this TMI issue is considered resolved for the ESBWR design.	
10 CFR 50. 34(f)(1)(viii)	II.K.3.21	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present (Applicable to BWR's only).	The comparable ESBWR systems are the ADS / GDCS and the ICS. The ADS is made up of SRVs and squib-activated DPVs. When the DPVs are actuated there is no way to manually stop depressurization and GDCS operation. Since the core is never uncovered the issues relating to spray cooling are not applicable to the ESBWR. This TMI item applies to low pressure inventory control systems (Core Spray and LPCI) that can be stopped by the operator. Once the ESBWR low pressure injection system, GDCS, is initiated, the operator does not have the ability to stop it from completing the initiation sequence. Therefore, this TMI item is not applicable to the ESBWR.	5.4.6, 6.3.2.7, 6.3.2.8
10 CFR 50. 34(f)(1)(ix)	II.K.3.24	Provide space cooling for RCIC, HPCI/HPCS systems for 2 hours following complete loss of offsite power. (Applicable to BWR's only).	The ESBWR ICS replaces the traditional HPCI and RCIC Systems found in most BWRs. The ICS does not rely on active pumps to remove excess sensible and core	5.4.6, 5.4.7,

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			decay heat. Each isolation condenser is located in a subcompartment of the IC/PCC pool, and requires no additional space cooling other than that provided by the surrounding water in the IC/PCC pool. If all of the safety-related power supplies used to start the ICs were to fail, then all available ICs automatically start into operation because of the "fail open" actuation of the condensate return bypass valves on loss of electrical power to the solenoids which control the pneumatically actuated valves. Therefore, this TMI item is considered resolved for the ESBWR design.	
10 CFR 50. 34(f)(1)(x)	II.K.3.28	Ensure that ADS valves, accumulators and associated equipment will be capable of performing its intended functions during and following an accident.	The ESBWR ADS is made up of SRVs and squib-activated DPVs. When the DPVs are actuated there is no way to close the DPVs until the valves are refurbished. The ADS utilizes the safety/relief valves (SRVs) and the depressurization valves (DPVs) for depressurization of the reactor. Each of the 10 ADS SRVs is equipped with a pneumatic accumulator and check valve for the ADS and manual opening functions. These accumulators assure that the valves can be opened following failure of the gas	5.2.2.2, 6.3.2.8, and 7.3.1.1.

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			supply to the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure, The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology. The DPVs are of a non-leak/non-simmer/non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The SRVs and DPVs and associated controls and actuation circuits are located or protected so that their function cannot be impaired by consequential effects of accidents. ADS components are qualified to withstand the harsh environments postulated for design basis accidents inside the containment, including high temperature, pressure, and radiation environments.	
10 CFR 50. 34(f)(1)(xi)	II.K.3.45	Ensure that vessel integrity limits are not exceeded during rapid depressurization and rapid cooldown.	The ESBWR ADS system DPVs are sized such that vessel depressurization and cooldown is slow enough that vessel integrity limits are not exceeded. A comprehensive thermal analysis was performed considering the effect of	5.3.2.1, 5.3.2.2, and 5.3.3.

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			blowdown and the Gravity-Driven Cooling System reflooding. Hypothetical ESBWR Accidents are much slower than those of previous BWR Product Line plants.	
10 CFR 50. 34(f)(1)(xii)	II.B.8	Include a hydrogen control system that satisfies the requirements of 10 CFR 50.34 (f)(2)(ix). As a minimum consider hydrogen ignition and post-accident inerting.	It is GE's position that this TMI item has been superseded by the revisions to 10 CFR 50.44. The ESBWR utilizes a nitrogen inerted containment to comply with this regulation and therefore complies with this TMI item.	6.2.5
10 CFR 50. 34(f)(2)(i)	I.A.4.2	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs	Simulator capability is not within the scope of ESBWR design certification. Defining the scope of simulator capability is the COL applicant's responsibility.	N/A
10 CFR 50. 34(f)(2)(ii)	I.C.9	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures.	Plant procedures are the responsibility of the COL applicant.	13.2.1, 13.3, 17.3, 18, and Appendices to Chapter 18.
10 CFR 50. 34(f)(2)(iii)	I.D.1	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts.	State-of-the-art human factor principles have been incorporated into the ESBWR control-room design. The design of the ESBWR control room utilizes accepted human factors engineering principles, incorporating the results of a full	18, 18D, 18E, and 18F.

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			systems analysis similar to that described in Appendix B of NUREG-0700. An integrated program plan, entitled “Design of Controls, Instrumentation and Man-Machine Interfaces,” was prepared and implemented to incorporate human factors engineering principles and to achieve an integrated design of the control and instrumentation systems and operator interfaces of the ESBWR. This plan and the associated procedures provided guidance for the conduct of the ESBWR control and instrumentation and Man-Machine Interface Systems (MMIS) design development activities including definition of the standard design features of the control room MMIS described in Subsection 18.4.2. Chapter 18 describes the ESBWR MMIS design goals and bases, the standard MMIS design features and the detailed MMIS design and implementation process, with embedded design acceptance criteria, for the ESBWR standard plant operator interface. A DCRDR specified in NUREG-0737 is not required by SRP Section 18.1.	
10 CFR 50.	I.D.2	Provide a plant safety parameter	The ESBWR Control Room Design	18.4.2.11

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
34(f)(2)(iv)		display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	incorporates the features that display to operators a set of parameters responding to the symptom driven EPGs defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. The principal functions of the SPDS as required by Supplement 1 to NUREG-0737 are integrated into the control room operator interface design, as permitted by SRP Section 18. The ESBWR control room operator interface design incorporates the SPDS function as part of the plant status summary information which is continuously displayed on the fixed-position displays on a large display panel, and also incorporates the use of on-screen control video display units (VDUs), independent of the plant computer, for control and monitoring of both safety-related and nonsafety-related systems. Other VDUs, driven by the plant computer, are available for monitoring of safety-related systems and monitoring and control of	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			nonsafety-related systems.	
10 CFR 50. 34(f)(2)(v)	I.D.3	Provide for automatic indication of the bypassed and [in]operable status of safety systems.	ESBWR design of I&C provides automatic indication of the bypasses and inoperable status of safety systems.	7.1.2.2, 7.2.1.3, and Table 7.1-1,
10 CFR 50. 34(f)(2)(vi)	II.B.1	Provide the capability of high point venting of noncondensable gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unaccepTable increase in the probability of loss-of-coolant accident or unaccepTable challenge to containment integrity.	During reactor operation the ESBWR design provides continuous venting from the RPV head and the IC driven by the differential pressure between the primary system pressure and a downstream steamline location where the non-condensables are extracted and swept to the main condenser, where the gasses are extracted. The capability to vent the ESBWR reactor coolant system when the vessel is isolated is provided by the safety/relief valves and reactor vessel head vent line. The head vent line is isolated from the Equipment and Floor Drainage System (EFDS) with two normally closed valves during reactor power operation. These vent and purge lines are not required to assure natural circulation core cooling.	5.4.6 and 5.4.12
10 CFR 50. 34(f)(2)(vii)	II.B.2	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an	The Alternate Source Term (AST) contained in Reg. Guide 1.183 has superseded the TID-14844 source term. The AST is used	3.1.2, 3.11,

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect equipment from the radiation environment.	for radiation design issues in the ESBWR. Reviews of ESBWR spaces requiring post-accident access reveals that each area has low post LOCA radiation levels. A review of the radiation and shielding of the ESBWR post-accident operations has been made. It has been found that there is adequate access to vital areas and that safety equipment is adequately protected. An evaluation of post-accident radioactive sources concluded that the ESBWR design limits potential radiation exposure from accidents both to plant personnel and to the public by the use of passive safety features and holdup in the containment. Potential releases in the radwaste building are contained by isolating the radwaste building atmosphere and containing any water releases in the building, which is seismically qualified and designed to prevent any potential water releases from high activity areas. Additional details relating to plant radiation sources can be found in Section 12.2. The locations requiring access to mitigate the consequences of an accident during the	12.3.5, 12.3.6, Figures 12.3-43 through 12.3-51, and 15.4.1.3.

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			post-accident period are the control room, the technical support center, the remote shutdown panel rooms, the primary containment sample station (process sample system), the health physics facility (counting room), the control room air bottles, the PCCS, ICS and fuel pool refill valves, and the nitrogen gas supply bottles. Each area has low post-LOCA radiation levels. The results of the evaluations are reflected in the radiation zone maps (Figures 12.3-43 through 12.3-51) and demonstrate that personnel doses will be within regulatory guidelines. The reactor building vital areas are all located off the controlled access way and contamination is limited to air infiltration from the containment and contaminated systems. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from monitoring systems such as the Process Sampling System (PSS). An environmental qualification program for safety-related mechanical and electrical equipment to demonstrate their capability to	

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			perform their required functions when exposed to the environmental conditions (including accident and post-accident conditions) in their respective locations is described in Section 3.11. Radiation shielding is designed to keep radiation doses to equipment below levels at which disabling radiation damage occurs	
10 CFR 50. 34(f)(2)(viii)	II.B.3	Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.	The ESBWR Containment Monitoring System (CMS) and Process Sampling System (PSS) provide the required capability to obtain and analyze highly radioactive post accident samples from the reactor coolant system, the containment sump, and the containment atmosphere. The Process Sampling System described in Subsection 9.3.2 meets the requirements of this position with the following exception. The upper limit of activity in the samples at the time they are taken is as follows: liquid sample 1 Ci/g; and gas sample 10^5 micro Ci/cc The PSS obtains reactor coolant and other samples following an accident. Liquid	7.5.2, 7.5.3, 9.3.2, and 11.5.

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			samples are taken from the Reactor Water Cleanup (RWCU) Inlets (hi-temp) and (lo-temp) via the connection to the Process Sampling System (PSS). The PSS collects representative liquid samples for analysis and provides the analytical information required to monitor plant and equipment performance and changes to operating parameters. The Containment Monitoring System (CMS) monitors the atmosphere in the containment for high gross gamma radiation levels and for high concentration levels of oxygen and hydrogen during post-accident conditions. Also, these three parameters are monitored during normal reactor operations to evaluate the integrity and safe conditions of the containment. Detailed descriptions of the PSS and CMS can be found in Subsections 9.3.2 and 7.5.2, respectively. Means to reduce radiation exposure are provided, such as shielding, remotely operated valves, and sample transporting casks. Acceptance Criterion II.K.5 of SRP Subsection 9.3.2 requires the capability of	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			sampling liquids of 10 Ci/g. The ESBWR design has the capability of sampling liquids of 1 Ci/g. Sampling and area radiation measurement would be performed. If levels are above safe limits, handling samples are delayed. The Process Radiation Monitoring System (PRMS) identifies the various gaseous and liquid process streams and effluent release paths or points to be monitored and sampled, and defines the required instrumentation for detection and measurement of the radioactive contents of these streams. The PRMS alerts operating personnel to excessive radiation levels and automatically initiates the required protection action to isolate radioactivity releases to the environs. The PRMS is designed for operability during and following an accident. A detailed description of the PRMS can be found in Subsection 7.5.3 and in Section 11.5.	
10 CFR 50. 34(f)(2)(ix)	II.B.8	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal	It is GE's position that this TMI item has been superseded by the revisions to 10 CFR 50.44. The ESBWR utilizes a nitrogen inerted containment to comply with	6.2.5

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide with reasonable assurance that: Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion. Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features. Equipment necessary for achieving and maintaining safe shutdown of the	this regulation and therefore complies with this TMI item.	

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system. If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.		
10 CFR 50. 34(f)(2)(x)	II.D.1	Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves … for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions shall be included.	The overpressure protection system, of which the SRVs are a part of, is capable of accommodating the most severe pressurization Anticipated Operational Occurrence (transient). The ESBWR pressurization is mild relative to previous BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization and does not result in opening of relief valves prior to	5.2.2.2, 6.3.2.8, and 7.3.1.1.

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			isolation condenser initiation. A detailed description of the safety evaluation of Anticipated Operational Occurrences (transients) for the overpressure protection system can be found in Subsection 5.2.2. The inspection and testing of applicable SRVs utilizes a quality assurance program, which complies with Appendix B of 10 CFR 50. The number for SRVs provided is sufficient to limit the pressurization of the RPV to less than code allowables in the event of an ATWS. The SRVs are tested at a suiTable test facility in accordance with quality control procedures to detect defects and to prove operability prior to installation. The conducted tests include hydrostatic, steam leakage, full flow pressure and blowdown, and response time testing. The valves are installed as received from the factory. The valve manufacturer certifies that design and performance requirements, including capacity and blowdown, have been met. The setpoints are adjusted, verified, and indicated on the valves by the	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			vendor. Specified manual and automatic initiated signal for power actuation of each the 10 ADS SRVs is verified during the preoperational test program described in Chapter 14. It is not feasible to test the SRV setpoints while the valves are in place. The valves can be removed for maintenance or bench testing and reinstalled during normal plant shutdowns. The valves will be tested to check set pressure in accordance with the requirements of the plant Technical Specifications. The external and flange seating surfaces of the SRVs are 100% visually inspected when any valve is removed for maintenance or bench testing during normal plant shutdown. The ESBWR ADS is made up of SRVs and squib-activated DPVs. When the DPVs are actuated there is no way to close the DPVs until the valves are refurbished.	
10 CFR 50. 34(f)(2)(xi)	II.D.3	Provide direct indication of relief and safety valve position (open or closed) in the control room.	Direct indication of SRV and DPV position (open or closed) is provided in the main control room. SRV position is indicated in the control room in full compliance with this	7.3.1.1

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			requirement.	
10 CFR 50. 34(f)(2)(xii)	II.E.1.2	Provide automatic and manual auxiliary feedwater (AFW) system initiation and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only)	Applicable to PWRs only. The ESBWR does not have an auxiliary feedwater system.	N/A
10 CFR 50. 34(f)(2)(xiii)	II.E.3.1	Provide pressurizer heater power supply and associated motive and control power … . (Applicable to PWRs only)	Applicable to PWRs only. The ESBWR does not have comparable systems.	N/A
10 CFR 50. 34(f)(2)(xiv)	II.E.4.2	Provide containment isolation systems that: (A) Ensure all non-essential systems are isolated automatically by the containment isolation system, (B) For each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with	The ESBWR Containment Isolation System meets the NRC requirements, including the post-TMI requirements. In general, this means that two barriers are provided. Redundancy and physical separation are required in the electrical and mechanical design of the containment isolation system to ensure that no single failure in the system prevents it from performing its intended functions. Electrical redundancy is provided for each set of isolation valves, eliminating dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line are routed separately.	3.1.5, 6.2.4, and 7.3.3

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		normal operations, and (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.	Cables are selected and based on the specific environment to which they may be subjected (e.g., magnetic fields, high radiation, high temperature and high humidity). Classification of structures, systems and components for the ESBWR design is addressed in Section 3.2 and identified in Table 3.2-1. The basis for classification is also presented in Section 3.2. The containment isolation system, in general, closes fluid penetrations for support systems that are not safety-related. The design of the control systems for automatic containment isolation valves ensures that resetting the isolation signal does not result in the automatic reopening of containment isolation valves. Actuation of the containment isolation system is automatically initiated by the Leak Detection and Isolation System (LD&IS) at specific limits defined for reactor plant operation. The LD&IS (described in Subsections 5.2.5 and 7.3.3) is designed to detect, monitor and alarm leakage inside and outside the containment, and automatically initiates the appropriate protective action to	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			isolate the source of the leak. Various plant variables are monitored, including pressure, and these are used in the logic to isolate the containment. The drywell pressure is monitored by four divisional channels, using pressure transmitters to sense the drywell atmospheric pressure from four separate locations. A pressure rise above the nominal level indicates a possible leak or loss of reactor coolant within the drywell. A high pressure indication is alarmed in the main control room, and initiates reactor scram and with the exception of the MSIVs, closure of the containment isolation valves. The alarm setpoints of the LD&IS are determined analytically or are based on actual measurements made during startup and pre-operational tests. All ESBWR containment purge valves meet the criteria provided in BTP CSB 6-4. The main purge valves are fail-closed and are verified to be closed at a frequency interval of 31 days as defined in the plant technical specifications. All purge and vent valves are pneumatically operated, fail closed and receive containment isolation signals. Bleed valves and makeup valves can be remote	

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			manually opened in the presence of an isolation signal, by utilizing override control if continued inerting is necessary. In the ESBWR design, redundant primary containment isolation valves (purge and vent) close automatically upon receipt of an isolation signal from the Leak Detection and Isolation System (LD&IS). The LD&IS is a four-divisional system designed to detect and monitor leakage from the reactor coolant pressure boundary, and, in certain cases, isolates the source of the leak by initiating closure of the appropriate containment isolation valves. Various plant variables are monitored, including radiation level, and these are used in the logic to initiate alarms and the required control signals for containment isolation. High radiation levels detected in the reactor building HVAC air exhaust or in the refueling area air exhaust automatically isolates the containment purge and vent isolation valves.	
10 CFR 50. 34(f)(2)(xv)	II.E.4.4	Provide a capability for containment purging/venting designed to minimize the purging time consistent	The ESBWR design includes a capability for containment purging/venting designed to minimize the purging time consistent with	7.3.3, and 9.4.9

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.	ALARA principles for occupational exposure. The system provides high assurance that the purge system will reliably isolate under accident conditions	
10 CFR 50. 34(f)(2)(xvi)	II.E.5.1	Establish a design criterion for the allowable number of actuation cycles of the ECCS and RPS consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only).	Applicable to B&W designs only. The ESBWR design includes criteria for the number of actuation cycles for the passive cooling components, which include both Anticipated Operational Occurrences (transients) and accidents.	3.9
10 CFR 50. 34(f)(2)(xvii)	II.F.1	Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite	The Containment Monitoring System (CMS) provides the ability to measure containment pressure, containment water level, containment hydrogen and oxygen levels, and radiation levels. Process Radiation Monitoring System (PRMS) monitors the radiation levels in gaseous streams at their release points.	7.5.2, and 7.5.3

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		capability to analyze and measure these samples.		
10 CFR 50. 34(f)(2)(xviii)	II.F.2	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as … a suiTable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's.	The detection of conditions indicative of inadequate core cooling is provided in the ESBWR design by the direct water level instrumentation system. Coolant level in the RPV is measured by both wide range and fuel zone instruments. The four divisions of wide range instruments cover the range from above the core to the main steam lines. The four channels of fuel zone instruments cover the range from below the core to the top of the steam separator. The RPV water level is the primary variable indicating the availability of adequate core cooling. Indication of water level by the differential pressure method is accepTable (without diverse methods of sensing and indication) because adequate redundancy and unambiguity is provided from the bottom of the core support plate to the centerline of the main steam lines. The ESBWR has addressed the issue regarding erroneously high water level indication upon vessel depressurization due to the release of	4.6 and 7.5.1

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			dissolved non-condensable gases in the reference leg. The ESBWR water level instrumentation system design includes a constant metered addition of purge water from the CRD hydraulic system to prevent the build-up of dissolved gasses in the fixed leg. This is consistent with the approved ABWR design as well as the modifications made by the majority of the BWR fleet.	
10 CFR 50. 34(f)(2)(xix)	II.F.3	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.	The ESBWR is designed in accordance with the most recent revision of Regulatory Guide 1.97, (Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident). A detailed assessment of the Regulatory Guide, including the list of instruments, is found in Section 7.5 of this DCD.	T 1.19-21, and 7.5.1
10 CFR 50. 34(f)(2)(xx)	II.G.1	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: … . (Applicable to PWRs only)	Applicable to PWRs only. The ESBWR does not have comparable design features to PWR pressurizers.	N/A
10 CFR 50. 34(f)(2)(xxi)	II.K.1.22	Design auxiliary heat removal systems such that necessary automatic and manual actions can be	There are no short term manual actions which must be taken. Sufficient systems exist to automatically mitigate the	7.2, 7.3, and 15.2.5.3

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only)	consequences of a loss of feedwater event. An analysis was performed for a loss of feedwater event. The sequence of events is described within Section 15.2.5.3, and is summarized below. In the event that the main feedwater system is not operable, a reactor scram and initiation of the ICS will occur either due to 1) A detected Loss of All Feedwater, or 2) Reactor water level will fall due to void collapse, boil-off and absence of makeup water. When Level 3 is reached, a reactor scram is automatically initiated. Reactor water level continues to decrease due to void collapse, boil-off until the low-low level setpoint, Level 2, is reached. At this point, reactor isolation also occurs, but with a time delay for the MSIVs. When an initiation signal is received by the isolation condensers (ICs), the condensate return valves will open in 30 seconds, placing the ICS in full operation at which time water level stabilizes. (High pressure CRD makeup if available will prevent the water level from falling to a point where ADS and GDCS are initiated.)	

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			If the ICs are not operable, the safety/relief valves (SRVs) will open on high vessel pressure approximately five minutes later. The SRVs open and close to maintain vessel pressure. When reactor low water Level 1.5 (immediate MSIV closure if not already closed) or 1 is reached (if ICs are not operable), an ADS timer is initiated. When the ADS timer is timed out, the ADS and Standby Liquid Control System (SLCS) actuation sequence is initiated, and the GDCS timer is initiated. When the GDCS timer is timed out, the GDCS injection valves open. Vessel pressure then decreases below the static head of GDCS, and the GDCS reflooding flow into the vessel begins. The core remains covered throughout the sequence of events and no core heatup occurs.	
10 CFR 50. 34(f)(2)(xxii)	II.K.2.9	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only).	The ESBWR does not have a system comparable to the B&W Integral Control System.	N/A

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
10 CFR 50. 34(f)(2)(xxiii)	II.K.2.10	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on the loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only).	The ESBWR Anticipated Operational Occurrences (transients) are generally much slower than even previous BWR designs. However, due to limited high pressure make-up, a reactor trip and initiation of the Isolation Condenser Systems (ICS) will occur in response to a Loss of All Feedwater event. But these are not anticipatory trips actuated directly on loss of main feedwater. The ESBWR includes as part of the reference design 110% bypass capacity for the main turbine. Scram only occurs on a turbine trip if an insufficient number of bypass open within a prescribed time period.	7.3, 7.4.4, 10.4.4.1, and 15.2.5.3.
10 CFR 50. 34(f)(2)(xxiv)	II.K.3.23	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWRs only).	Recording of water level is included in the MCR. Water level measurements are from the wide and fuel range water level instruments. See the discussion of 34(f)(2)(xvii) for more detail.	7.5.1
10 CFR 50. 34(f)(2)(xxv)	III.A.1.2	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near-site Emergency Operations Facility.	Space for the Technical Support Center is included in the Standard Design on the ground floor of the Electrical Building. The space provided is in conformance with NUREG-0696	Figure 1.2-26, and 13.3

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			The COL applicant will address provisions for an onsite Operational Support Center, and a near-site Emergency Operations Facility.	
10 CFR 50. 34(f)(2)(xxvi)	III.D.1.1	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	The Leak Detection and Isolation System (LD&IS) includes detection and actuation capability for systems that could potentially carry radioactive material outside the containment. Containment integrated leakage rate (Type A tests), containment penetration leakage rates (Type B tests), and containment isolation valve leakage rates (Type C tests) that comply with Appendix J and General Design Criteria 52, 53, and 54 of Appendix A of 10 CFR 50. Type A, B, and C tests are performed prior to operations and periodically thereafter to assure that leakage rates through the containment and through systems or components that penetrate the containment do not exceed maximum allowable rates specified in the plant Technical Specifications (TS). There are 7 systems that could contain radioactive material outside the primary containment.	6.2.6, and 7.3.3

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
			1) Passive Containment Cooling System (is considered part of the containment boundary) 2) Isolation Condenser System 3) Reactor Water Cleanup System/Shutdown Cooling 4) Main Steam System 5) Fuel and Auxiliary Pool Cooling System 6) Containment Inerting System 7) Equipment and Floor Drainage System (Lower Drywell Sumps)	
10 CFR 50. 34(f)(2)(xvii)	III.D.3.3	Provide for monitoring of inplant radiation and airborne radioactively as appropriate for a broad range of routine and accident conditions.	The ESBWR provides three systems to monitor area radiation and airborne radioactivity: Containment Monitoring System (CMS), Process Radiation Monitoring System (PRMS) Area Radiation Monitoring System (ARMS).	7.5.2, 7.5.3, and 7.5.4
10 CFR 50. 34(f)(2)(xxviii)	III.D.3.4	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term	Safe occupancy of the ESBWR control room during abnormal conditions is provided for in the design. Adequate shielding is provided to maintain tolerable radiation levels in the control room in the event of a	3.1.2, 6.4.2, 9.4.1, and 15.4

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		release, and make necessary design provisions to preclude such problems.	design basis accident for the duration of the accident. The Control Room Area Ventilation System has redundant equipment and includes HEPA and Charcoal filters and radiation detectors with appropriate alarms and interlocks. If any hazards exist at the normal control room ventilation intake, habitability is assured by the Emergency Breathing Air System (EBAS), which upon isolation of the control room habitability area, provides a positive air purge. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided outside the control room, which can be utilized to initiate reactor shutdown, maintain a safe shutdown condition and achieve subsequent cold shutdown of the reactor.	
10 CFR 50. 34(f)(3)(i)	I.C.5	Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those	The ESBWR design engineers are continually involved in reviewing industry experience from sources such as NRC Bulletins, Licensee Event Reports, NRC request for information letters to holders of operating licenses for nuclear power	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		designing and constructing the plant.	reactors, Federal Register information, and generic letters.	
10 CFR 50. 34(f)(3)(ii)	I.F.1	Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR Part 50 includes all structures, systems, and components important to safety.	The ESBWR Quality Assurance Plan is described in Chapter 17. Structures, systems, and components are classified as described in Section 3.2.	3.2, and 17
10 CFR 50. 34(f)(3)(iii)	I.F.2	Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff	The ESBWR Quality Assurance Plan described in Chapter 17 meets the requirements of issue I.F.2 as they apply to the design of the ESBWR.	17

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities.		

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
10 CFR 50. 34(f)(3)(iv)	II.B.8	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.	The Containment Inerting System provides the capability to vent the containment via a pathway that connects the wetwell airspace to the stack. This pathway can be opened in the event that the operators determine that venting is required and provides a fission product release at an elevated point at a time prior to containment structural failure. Having the connection point in the wetwell airspace forces the escaping fission products through the suppression pool. In a core damage event initiated by an Anticipated Operational Occurrence (transient), in which the vessel does not fail, fission products are directed to the suppression pool via the SRVs, DPVs via the connecting vents, ICS or PCCS, scrubbing any potential release. Therefore there is no need for any dedicated penetrations to be provided.	9.4.9 and Figure 9.4-14

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
10 CFR 50. 34(f)(3)(v)	II.B.8	Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8) (A)(1) Containment integrity will be maintained (i.e., for steel containment … . For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an	The ESBWR has a concrete containment that meets the requirements of this provision. Compliance with Reg. Guide 1.7 demonstrates that this issue has been satisfactorily addressed. See further detailed discussion in the response to 10 CFR 50.34(f)(2)(ix).	6.2.5

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions. Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)9v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. (B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not		

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category, (2). The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.		
10 CFR 50. 34(f)(3)(vi)	II.E.4.1	For plant designs with external hydrogen recombiners, provide redundant dedicated containment	The ESBWR does not have <u>external</u> hydrogen recombiners, therefore, this requirement is not applicable.	N/A

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.		
10 CFR 50.34 (f)(3)(vii)	II.J.3.1	Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the	The ESBWR design team has developed a management plan for the ESBWR project which consists of a properly structured organization with open lines of communication, clearly defined responsibilities, well-coordinated technical efforts, and appropriate control channels. The procedures to be used in the construction, startup, and operation phases of the plant are provided by the Combined Operating License (COL) applicant.	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Tier 2 Location(s)
		preparation and implementation of procedures necessary to guide the effort.		

APPENDIX 1B PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECTIVE SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION [II.B.2]

1B.1 INTRODUCTION

General Electric has performed a review of the ESBWR post-accident environment in response to NUREG-0737 Item II.B.2. This attachment discusses the results of that review.

1B.2 SUMMARY OF SHIELDING DESIGN REVIEW

Several alternatives are potentially available to the designer to assure continued equipment availability and performance under post-accident conditions. One alternative is to provide redundant systems and/or components, which are qualified to operate in the expected environment and thus preclude the need for operator access. Another is to provide operator access to conduct the operations and to maintain the equipment. This latter alternative would generally be accompanied by appropriate shielding and, in many cases, would be difficult if not impossible to carry out.

GE has taken the first approach and furthermore has designed the plant so that most responses to transient conditions are automatic, including achieving and maintaining safe-shutdown conditions. The design basis for the ESBWR is to require safety-related equipment to be appropriately environmentally qualified and operable from the control room. As a result of this design philosophy and as shown by this review, no changes are necessary to assure that personnel access is adequate or that safety equipment is not degraded because of post-accident operation.

As part of the design of the ESBWR, it was necessary to establish the environmental conditions for qualification of safety-related equipment. A result of this design work was an environmental requirement establishing the integrated dose that the equipment must be able to withstand. These values are listed in Appendix 3H.

Another aspect of the review was the manner in which the safety-related equipment is arranged and operated during normal and abnormal operation and postulated accidents. The essence of the ESBWR is to achieve and maintain a safe shutdown condition for all postulated accident conditions with operator actions being conducted from outside the containment zones, principally from the control room.

The purpose of this review is first to verify that, where equipment access is required, it is reasonably accessible outside the containment zones. Secondly, the review should verify that inaccessible equipment is environmentally qualified and is operable from the control room.

The results of the review are:

(1) The period of interest begins with the plant in a safe shutdown condition. Thus, the various safety-related systems needed to achieve safe shutdown conditions have performed as expected, and only the engineered safety features systems (Chapter 6) and auxiliaries, as described later, are required to maintain this condition.

(2) Based upon the accident source terms of Regulatory Guides 1.183 and 1.7 and Standard Review Plan 15.0.1 including normal operations, the vital equipment exposures are enveloped based upon the Table below:

Area	Gamma (Gy)	Beta (Gy)
Containment	$2x10^6$	$2x10^7$

Each actual area is environmentally qualified to the area specific envelope as defined in Appendix 3H.

(3) It is not necessary for operating personnel to have access to any place other than the control room, technical support center, post-accident sampling station, sample analysis area, and nitrogen supply bottles to operate the equipment of interest during the 100-day period. The control room, technical center and sample analysis area are designed to be accessible post-accident. The latter areas are considered accessible on a controlled exposure basis.

(4) Access to radwaste is not required, but the Radwaste Building (RW) is accessible, since containment sump discharges are isolated. Thus, fission products are not transported to radwaste. The ESBWR does not have a containment isolation reset control area. These functions are provided in the control room.

(5) Following an accident, access is available to electrical equipment rooms containing motor control centers and corridors in the upper RW (see post-accident radiation zone maps in Subsection 12.3.6). This is based on radiation shine from the containment. While not necessary to maintain safe shutdown, such access can be useful in extending system functionality and plant recovery.

(6) The safety-related power supplies identified in Table 1B-5 are accessible. However, access is not necessary. Nonsafety-related diesel generators are also available and accessible to provide power.

1B.3 CONTAINMENT DESCRIPTION AND POST-ACCIDENT OPERATIONS

1B.3.1 Description of Containment

The ESBWR design includes many features to assure that personnel occupancy is not unduly limited and that safety-related equipment is not degraded by post-accident radiation fields. These features are detailed in Tier 2. Consequently, only a brief summary description and Tier 2 reference are provided here for emphasis.

The configuration of the pressure suppression containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the containment atmosphere following an accident is thereby substantially reduced compared to the Reg. Guide 1.183 source terms. The Passive Containment Cooling System (PCCS) condensing function contributes to reduce many of the airborne fission products.

Containment leakage is limited to less than one half percent of the containment atmosphere per day.

Radiation to the Reactor Building is limited to shine through the walls. There is no airborne radiation in these other areas. As these become accessible after an accident, any component failures can be repaired, thereby improving systems availability.

1B.3.2 Post-Accident Access of Vital Areas and Systems

A vital area is any area that may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas that must be considered as vital after an accident are the control room, technical support center, sampling station and sample analysis area. The nonsafety-related High Pressure Nitrogen Supply System (HPNSS) nitrogen supply bottles are available for use to operate containment isolation valves inside containment if necessary in support of long-term post-accident actions.

The vital areas also include consideration (in accordance with NUREG-0737, II.B.2) of the containment isolation reset control area, manual ECCS alignment area, motor control center and radwaste control panels. However, the ESBWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room. Those vital areas, which are normally areas of mild environment allowing unlimited access, are not reviewed for access.

Essential systems specific to the ESBWR to be considered post-accident are those for long-term core cooling, fission product control and combustible gas monitoring, as well as the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1B.3.3 Post-Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10 CFR 50.34(a) guidelines.

Safety-related systems are required for scram and to achieve a safe shutdown condition. However, they are not necessarily needed to maintain safe shutdown. The systems considered herein are the safety-related engineered safety features (ESF) (Chapter 6) used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the post-accident sampling station, the sample analysis area, and manual nitrogen reserve supply valves.

1B.4 DESIGN REVIEW BASES

1B.4.1 Radioactive Source Term and Dose Rates

The radioactive source term used is equivalent to the source terms recommended in Reg. Guide 1.183 and Standard Review Plan 15.0.1 with appropriate decay times. Depressurized coolant is assumed to contain no noble gas.

Dose rates for areas requiring continuous occupancy may be averaged over 30 days to achieve the desired <0.15 mSv/h.

Design dose rates for personnel in a vital area are such that the guidelines of General Design Criterion (GDC) 19 (i.e., <0.05 Sv) total effective dose equivalent or its equivalent to any part of the body are not exceeded for the duration of the accident, based upon expected occupancy and protection.

1B.4.2 Accidents Used as the Basis for the Specified Radioactivity Release

The various accidents and associated potential for fuel rod failure are addressed in Sections 15.3 and 15.4. This chapter also provides the accident parameters. Of those accidents, only the design basis accident (DBA) LOCA is assumed to produce 100% failed fuel rods under NRC worst-case assumptions. The fuel handling accident is the only other DBA postulated as leading to failed fuel rods with the potential consequence of radioactivity releases comparable to the 10 CFR 50.34 (a) guidelines.

For the fuel handling accident, the reactor is either shutdown and cooled or is operating normally if the accident is in the spent fuel storage pool. The total activity released to the environment and the calculated exposures are provided in Subsection 15.4.1. The exposures are within the guidelines of 10 CFR 50.34(a). Thus, recovery is possible well within the specified 100-day equipment qualification period. ECCS equipment is not affected by this accident. This accident is not considered further.

Although a DBA-LOCA would not actually uncover the core or lead to fuel damage (see Section 6.3), core wide fuel failure is assumed such that this accident produces the limiting conditions of interest for this design review. In this accident the reactor is depressurized and reactor water mixes with suppression pool water in the process of keeping the fuel covered and cooled.

1B.4.3 Availability of Offsite Power

The availability of offsite power is not influenced by plant accident conditions. Loss of offsite power may be assumed as occurring coincident with the beginning of the accident sequence. However, continued absence of offsite power for the accident duration is not realistic. While restoration of offsite power is not a necessary condition for maintaining core cooling, its availability can permit operation of other plant systems that would not otherwise be permitted by emergency power restrictions (e.g., operation of the pneumatic air system, nonsafety-related HVAC systems and other systems useful to plant cleanup and recovery).

Based on the PRA in Chapter 19, the probability for offsite power recovery is estimated to be very high in 8 hours, while the AC power is not needed for at least 72 hours following an accident.

1B.4.4 Radiation Qualification Conditions

The safety-related equipment requiring review for qualification is only that necessary for post-accident operations and for providing information for assuring post-accident control.

In 10 CFR 50.46, the long-term cooling capability is given as follows: "...decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core." A 100-day period has been selected as a sufficient extended period permitting site and facility response to terminate the event.

As part of the design review process, a set of reference conditions is necessary for comparing expected post-accident radiation exposures. Appendix 3H defines the environmental conditions for safety-related equipment zones for periods of 60 years normal operations, including anticipated tests and abnormal events, and six months following the DBA-LOCA. These conditions are upper bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. In effect, these are specification values, and equipment is qualified to meet or exceed these values.

Radiation sources in the containment are the same as the Table 1B-1 design basis values for water sources. For airborne radiation sources, the plant design basis of Table 1B-1 for air is used. Containment leakage is assumed to occur in each of the individual Reactor Building compartments. As previously noted, no credit has been taken for the radio-halogen scrubbing, which is an inherent feature of the BWR.

1B.5 RESULTS OF THE REVIEW

1B.5.1 Systems Required Post-Accident

This section establishes the various equipment required to function following an accident along with their locations. The expected habitability conditions and access and control needs are identified for the required post-accident period.

1B.5.1.1 Necessary Post-Accident Functions and Systems

Following an accident and assuming that immediate plant recovery is not possible, the following functions* are necessary:

(1) Reactivity control

(2) Reactor core cooling

(3) Reactor coolant pressure boundary integrity (if not already breached by the initiating event)

(4) Containment integrity

(5) Radioactive effluent control

Reactivity control is a short-term function and is achieved when the reactor is shutdown. The remaining functions are achieved in the longer-term post-accident period by use of:

(1) The Emergency Core Cooling System (ECCS) (for reactor core cooling);

(2) The Passive Containment Cooling System (PCCS) (for containment heat removal);

(3) The fission product removal and control system and auxiliaries (for radioactive effluent control);

(4) Instrumentation and controls and power for accident monitoring and functioning of the necessary systems and associated habitability systems.

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* ANSI/ANS 4.5 Criteria for Accident Monitoring Functions in Light Water Reactors.

Tables 1B-2 through 1B-5 are generated to show:

- What major equipment and systems are required to function and thereby define the systems for review; and
- The redundant equipment locations by divisional isolated room or area and containment or building.

1B.5.1.2 Emergency Core Cooling and Residual Heat Removal Systems

The Gravity-Driven Cooling System (GDCS) provides cooling for the fuel under accident conditions as described in Subsection 6.3.2.7. After it is initiated the GDCS requires no auxiliary support or post-accident instrumentation.

The Automatic Depressurization System (ADS) function is described in Subsection 6.3.2.8. A postulated small break accident could require the depressurization function until the GDCS is initiated. In the case of a small break accident, the majority of the fission products would be released via the safety/relief valves to the suppression pool and hence to the containment, rather than direct mixing through the suppression pool vents, as would occur following a DBA-LOCA. In either case, the distribution of fission products is assumed the same as for the DBA-LOCA even though, realistically, a significant portion of halogens and solid fission products would be retained in the reactor pressure vessel. Thus, the results as they apply to the ADS are conservative. The pneumatic nitrogen supply for the ADS is supplied by the SRV accumulators included in Table 1B-2. The pneumatic nitrogen supply for other containment valves is included in Table 1B-3 as a portion of the containment auxiliaries. The hand-operated nitrogen reserve supply valves are accessible outside the containment, if needed, to mitigate a large nitrogen leak.

Credit is also taken during a LOCA for the water volume stored in the discharge lines of the Isolation Condenser System (ICS) and for operation of the Standby Liquid Control (SLC) system.

Containment cooling is provided by the PCCS. PCCS is a passive system that requires no operator action. The PCCS function cools the air volumes and discharges the resulting condensate to the GDCS pools so that it is available for return to the reactor pressure vessel. Additional containment cooling can be obtained by manually initiating the nonsafety-related Fuel and Auxiliary Pools Cooling System (FAPCS) in its drywell spray mode. Controls for initiating drywell spray are available in the main control room.

The fuel pool cooling function is also provided in the event that a recently unloaded fuel batch requires continued cooling during the post-accident period. The spent fuel pool contains sufficient inventory to ensure no operator action is required during the first 72 hours. After that period, either makeup water must be supplied to the spent fuel pool or the FAPCS must be initiated. The FAPCS equipment is environmentally qualified, so access is not required and redundancy is included in system components.

The locations of selected ECCS equipment and instrument transmitters are included in Table 1B-2. These listings do not represent all the types of this equipment that are environmentally qualified, safety-related, or included in the systems classified in Table 3.2-1. It does, however, represent principal components that are needed to operate, generally during post-accident operations. For example, (after depressurization) only the GDCS discharge valves need to open to direct water to the reactor. Similarly, the instrument transmitters shown are those that would provide information for system initiation and monitoring of long-term system performance post-accident. Control room instrumentation is not listed, because it is in an accessible area where no irradiation degradation would be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under the Nuclear Boiler System (B21) are those for ECCS functions or monitoring reactor vessel level.

1B.5.1.3 Flammability Control

Flammability control in the containment is achieved by an inert atmosphere during all plant operating modes except operator access for refueling and maintenance. The Containment Inerting System gas supply is described within Subsection 9.4.9. The Containment Monitoring System (CMS) measures and records oxygen and hydrogen concentrations in the containment under post-accident conditions. It is automatically initiated by detection of a LOCA (Subsection 7.5.2). Table 1B-3 lists the principal combustible gas monitoring components and their locations.

1B.5.1.4 Fission Product Removal and Control System

The ESBWR does not need, and thus does not have, a filter system that performs a safety-related function following a design basis accident. The control room is provided with self contained bottled air to maintain a safe control room atmosphere following a design basis accident as discussed in Section 6.4.

The CMS described in the previous section also measures and records containment area radiation under post-accident conditions. A post-accident sampling subsystem (PASS) obtains containment atmosphere and reactor water samples for chemical and radiochemical analysis in the laboratory. Delayed sampling, shielding, remote operated valves and sample transporting casks are utilized to reduce radiation exposure. The samples are manually transported between the PASS room in the Reactor Building and the analysis laboratory in the Service Building. The system is described in Subsection 9.3.2. Table 1B-4 lists the fission product removal control components and locations.

1B.5.1.5 Instrumentation and Control, Power and Habitability Systems

Most of the post-accident instrumentation and control system equipment is listed with the applicable equipment in Tables 1B-2, 1B-3 and 1B-4. The remaining instrumentation and control equipment is included with the power and habitability systems equipment listed in Table 1B-5. Instrumentation is consistent with the post-accident phase variables monitored by the post-accident monitoring functions of the CMS (see Table 7.5-2).

The ESBWR does not need/have safety-related standby diesel generators. Storage batteries are the standby power source for Class 1E electric power.

Habitability systems ensure that the operator can remain in the control room and take appropriate action for post-accident operations. The Control Building includes the instrumentation and controls necessary for operating the systems required under post-accident conditions.

Table 1B-1

Radiation Source Comparison

Activity Group	**% Core Inventory Released**		
	Reg. Guide 1.183	Reg. Guide 1.7	Plant Design Basis
Air			
Noble Gases	100	100	100*
Halogens	30	—	30*
All Remaining	—	—	—
Water			
Noble Gases	0	—	0
Halogens	—	50	50**
All Remaining	—	1	1

* Uniformly mixed within the containment boundary
** Uniformly mixed in the suppression pool and reactor coolant

Table 1B-2

Post-Accident Emergency Core Cooling Systems and Auxiliaries

Equipment	MPL	Location
ADS		
Safety Relief Valves	B21	Upper Drywell (C)
SRV Accumulators	B21	Upper Drywell (C)
Depressurization Valves	B21	Upper Drywell (C)
GDCS		
Discharge Valves	E50	Upper Drywell (C)
Isolation Condensers		
Steam Supply Valves	B32	Upper Drywell (C)
Condensate Discharge Valves	B32	Upper Drywell (C)
Condenser Units	B32	IC/PCC Pools (C)
Standby Liquid Control (SLC) system		
Inboard Isolation Valves	C41	Upper Drywell (C)
Outboard Isolation Valves	C41	SLC Rooms (RB)
Accumulators	C41	SLC Rooms (RB)
Initiation / Monitoring Instrumentation		
Reactor Water Level	B21	Instrument Rack Room (RB)
Reactor Pressure	B21	Instrument Rack Room (RB)
Drywell Pressure	T62	Instrument Rack Room (RB)
Wetwell Pressure	T62	Instrument Rack Room (RB)

(C) — Containment

(RB) — Reactor Building

Table 1B-3

Post-Accident Containment Monitoring and Auxiliary Systems

Equipment	MPL	Location
High Pressure Nitrogen Supply System (HPNSS)		
Nitrogen Storage Bottles	P54	By Valve Room (RB)
Supply Pressure	P54	By Valve Room (RB)
Containment Monitoring System (CMS)		
Hydrogen, Oxygen Elements	T62	CMS Rooms (RB)
Gas Measurement	T62	CMS Rooms (RB)
Gas Elements	T62	CMS Rooms (RB)
Drywell Gas Valve	T62	CMS Rooms (RB)
Wetwell Gas Valve	T62	CMS Rooms (RB)
Gas Supply	T62	CMS Rooms (RB)

(C) — Containment

(RB) — Reactor Building

Table 1B-4

Post-Accident Fission Product Removal and Control Systems and Auxiliaries

Equipment	MPL	Location
Emergency Breathing Air System		
	U65	(EBAS)
Post-Accident Sampling Subsystem (PASS)		
Conditioning/Holding Rack	P33	(RB)
Sampling/Casks Rack	P33	PASS Rack Rm. (RB)
DW/WW Gas (CMS) Valve	T62	(RB)
Control Panel	H21	PASS Rack Rm. (RB)
Chemical Radiological Analysis	COL applicant information	Laboratory (SB)
Stack		
Radiation (Ion/Scint.)	D11	Stack

(CB) — Control Building

(RB) — Reactor Building

(SB) — Service Building

(EBAS) — Emergency Breathing Air System Building

Table 1B-5

Post-Accident Instrumentation and Controls, Power and Habitability Systems and Auxiliaries

Equipment	MPL	Location
Instrumentation and Controls		
Post-Accident I&C	H11-Post-Accident	Control & Panel Rooms (CB)
Distributed Control and Information System	C63	
Power		
DC Supply	R16-Storage Batteries	(RB)
AC Low Voltage and I&C Supply Systems	R13, R14-Post-Accident	(RB)
Control Building HVAC		
Detection of high airborne radioactivity, toxic gases or smoke	D21	(CB)
Isolation of Sealed Emergency Operating Area	U77	(CB)
Emergency Breathing Air controls	U77	(CB)

(RB)—Reactor Building

(CB)—Control Building

APPENDIX 1C INDUSTRY OPERATING EXPERIENCE

1C.1 EVALUATION

Industry operating experience information is routinely made available to and/or distributed by GE design and modifications personnel. The more important industry-wide issues are routinely addressed in NRC Generic Letters and Bulletins.

All of the Generic Letters and Bulletins covering January 1, 1980 through February 24, 2005 were reviewed. Of those, the Generic Letters and Bulletins that are potentially applicable to the ESBWR design or operations are addressed in Table 1C-1. For each of those Generic Letters and Bulletins, Table 1C-1 provides (a) the Tier 2 location(s) where the Generic Letter's or Bulletin's topic is addressed, (b) a summary conclusion of its effect on the ESBWR, or (c) notes that it is applicable to the COL applicant/holder. (See Subsection 1.9.4 for COL information.). Generic Letter and Bulletin topics deemed not applicable to the ESBWR design or operations (topics pertaining to other reactor types or BWR design features, e.g., a Reactor Recirculation Pump issue) are not included in Table 1C-1. Also, Generic Letter and Bulletin topics related to identified regulatory or industry developed resolutions are not included in Table 1C-1, to avoid repetition within Tier 2.

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
Type: Generic Letters			
80-06	4/25/80	Clarification of NRC Requirement for Emergency Response Facilities at Each Site	The ESBWR includes provisions for a Technical Support Center (TSC). The Operational Support Center (OSC) and Emergency Operating Facility (EOF) are the responsibility of the COL Applicant. Section 13.3.
80-30	12/15/80	Periodic Updating of Final Safety Analysis Reports (FSARs)	COL Holder
80-34	4/25/80	Clarification of NRC Requirements for Emergency Response Facilities at Each Site	See response to GL 80-06.
80-113	12/22/80	Control of Heavy Loads	Subsection 9.1.5 with COL Holder to supplement
81-03	2/26/81	Implementation of NUREG-0313m, Rev. 1	Subsection 5.2.3
81-04	2/25/81	Emergency Procedures and Training for Station Blackout Events	COL Holder
81-07	2/3/81	Control of Heavy Loads	Subsection 9.1.5 with COL Holder to supplement
81-10	2/18/81	Post-TMI Requirements for the Emergency Operations Facility	See response to GL 80-06.
81-11	2/22/81	Comments on NUREG-0619	Not applicable. The ESBWR does not include a CRD return nozzle on the RPV.
81-20	4/1/81	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Not applicable. The ESBWR utilizes FMCRDs and does not have CRD withdraw lines and scram discharge volume.
81-37	12/29/81	ODYN Code Reanalysis Requirements	Not applicable, ESBWR does not use the ODYN Code

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
81-38	11/10/81	Storage of Low-Level Radioactive Wastes at Power Reactor Sites	The Radwaste Building includes space for processing and storage of low level waste. Storage space is provided for 6 months worth of waste. Section 11.4 with COL Applicant to supplement
82-09	4/20/82	Environmental Qualification of Safety-Related Electrical Equipment	Section 3.11
82-21	10/6/82	Technical Specifications for Fire Protection Audits	Not applicable. No longer controlled by Technical Specifications.
82-23	10/30/82	Inconsistency Between Requirements of 10CFR73.40(d) and Standard Technical Specifications for Performing Audits of Safeguard Contingency Plans	Not applicable. No longer controlled by Technical Specifications.
82-27	11/15/82	Transmittal of NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants," and NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."	The suppression pool is provided with sufficient temperature instrumentation to monitor the temperature rise during testing that adds heat to the pool. The Tech Spec limits on pool temperature are established in accordance with the applicable design limits. Subsection 7.2.3.
82-33	12/17/82	Supplement 1 to NUREG-0737	These requirements have been incorporated into Reg. Guide 1.97 and the ESBWR conforms with this Reg. Guide. Appendix 1A
82-39	12/22/82	Problems with the Submittals of 10 CFR 73.21 Safeguards Information Licensing Review	GE has an approved procedure for handling Safeguards Information including how to mail such information to authorized recipients.
83-05	2/8/83	Safety Evaluation of "Emergency Procedure Guidelines," Revision 2, NEDO-24934, June 1982	Appendix 18A
83-07	2/16/83	The Nuclear Waste Policy Act of 1982	COL Applicant

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
83-13	3/2/83	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems	Subsection 9.4.1
83-28	7/8/83	Required Actions Based on Generic Implications of Salem ATWS Events	Superseded by 10 CFR 50.62, see Section 15.5 for the ATWS event evaluation
83-33	10/19/83	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	Subsection 9.5.1 and Appendices 9A & 9B
84-15	7/2/84	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	Not applicable, the ESBWR does not have/need emergency diesel generators.
84-23	10/26/84	Reactor Vessel/Water Level Instrumentation in BWRs	Subsection 7.7.1
85-01	1/9/85	Fire Protection Policy Steering Committee Report	Subsection 9.5.1 and Appendices 9A & 9B
86-10	4/24/86	Implementation of Fire Protection Requirements	Subsection 9.5.1 and Appendices 9A & 9B
87-06	3/13/87	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	Not applicable, as defined Section B3.4.6 of NUREG-1434, the ESBWR does not need nor have pressure isolation valves.
87-09	6/4-87	Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operations and Surveillance Requirements	DCD Chapter 16 TS Section 3.0, consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.0).
88-01	1/25/88	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping	Subsection 5.2.3.4
88-14	8/8/88	Instrument Air Supply System Problems Affecting Safety-Related Equipment Past Related Correspondence: IE Notice 87-28, Supp. 1 NUREG-1275, Volume 2	Subsection 9.3.6

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
88-15	9/12/88	Electric Power Systems— Inadequate Control Over Design Process Past Related Correspondence: IE Notice 88-45	Most of the issues described in GL88-15 are not applicable to the ESBWR. Section 8.3
88-16	10/4/88	Removal of Cycle-Specific Parameter Limits from Technical Specifications	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.0). Subsection 16.5.6.3
88-18	10/20/88	Plant Record Storage on Optical Disks Past Related Correspondence: NUREG-0800 Reg. Guide 1.28, Rev. 3	COL Applicant and Holder to supplement Subsection 17.1.17
88-20	11/23/88	Individual Plant Examination for Severe Accident Vulnerabilities-10CFR Para. 50.54(f)	Chapter 19
88-20s1	8/29/89	Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities-10CFR Para. 50.54(f)	Chapter 19
88-20s2	4/4/90	Accident Management Strategies for Consideration in the Individual Plant Examination Process	Chapter 19
88-20s3	7/6/90	Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities	Chapter 19
88-20s4	6/28/91	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)	Chapter 19
88-20s5	9/8/95	Individual Plant Examination of External Events for Severe Accident Vulnerabilities	Chapter 19

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
89-01	1/31/89	Implementation of programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.0). COL Applicant/Holder Subsections 16.5.5.1 and 16.5.5.3,
89-02	3/21/89	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products Past Related Correspondence: EPRI-NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications". Bulletins 87-02 and Supplements 1 and 2, 88-05 and Supplements 1 and 2, 88-10 IE Notices 87-66, 88-19, 88-35, 88-46 and Supplements 1 and 2, 88-48 and Supplement 1, 88-97	COL Applicant/Holder
89-04	4/3/89	Guidance on Developing AccepTable Inservice Testing Program	COL Holder to supplement Subsections 3.9.6, 5.2.4, 6.3.3.9, and 6.6
89-04s1	4/4/95	Guidance on Developing AccepTable Inservice Testing Programs	Subsection 5.2.4 and Section 6.6
89-06	4/12/89	Task Action Plan Item I.D.2 – Safety Parameter Display System CFR 50. 34(f)	Appendix 1A and Chapter 18
89-07	4/28/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	COL Applicant/Holder
89-07 Supp I	4/21/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	COL Applicant/Holder
89-08	5/2/89	Erosion/Corrosion-Induced Pipe Wall Thinning	Subsection 6.6.7
89-10	6/28/89	Safety-Related Motor-Operated Valve Testing and Surveillance	COL Applicant/Holder

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
89-10s1	6/13/90	Results of Public Workshop	COL Applicant/Holder
89-10s3	10/25/90	Consideration of the Results of NRC Sponsored Tests of Motor-Operated Valves	COL Applicant/Holder
89-10s4	2/12/92	Consideration of Valve Mispositioning in Boiling Water Reactors	COL Applicant/Holder
89-10s5	6/28/93	Inaccuracy of Motor-Operated Valve Diagnostic Equipment	COL Applicant/Holder
89-10s6	3/8/94	Information on Schedule and Grouping, and Staff Responses to Additional Public Questions	COL Applicant/Holder
89-13	7/18/89	Service Water System Problems Affecting Safety-Related Equipment	Not applicable, ESBWR has no safety-related service water
89-13s1	4/4/90	Service Water System Problems Affecting Safety-Related Equipment	Not applicable, ESBWR has no safety-related service water
89-14	8/21/89	Line Item Improvements in Technical Specifications Removal of the 3.25 Limit on Extending Surveillance Intervals	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.0) Section 16.3
89-15	8/21/89	Emergency Response Data System	COL Applicant
89-16	9/1/89	Installation of a Hardened Wetwell Vent	The ESBWR does not need a dedicated Hardened Wetwell Vent, as discussed in Subsection 6.2.5.4
89-18	9/6/89	Resolution of USI A-17, Systems Interactions	Section 1.11 and 19
89-19	9/20/89	Request for Action Related to Resolution of Unresolved Safety Issue A-47, "Safety Implication of Control Systems in LWR Nuclear Power Plants", Pursuant to 10 CFR 50.54(f)	The ESBWR utilizes a common adjusTable speed drive motor for both the FW and condensate booster pumps. Section 1.11, Subsection 7.7.3

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
89-22	10/19/89	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed By The National Weather Service	Section 2.3
90-09	12/11/90	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	COL Holder
91-03	03/06/91	Reporting of Safeguards Events	COL Applicant/Holder
91-04	04/02/91	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	DCD Chapter 16 TS Sections 3.1-3.9, COL Holder scope
91-05	04/04/91	Licensee Commercial Grade Procurement and Dedication Programs	COL Holder
91-06	04/29/91	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies", Pursuant to 10 CFR 50.54(f)	Subsection 8.3.2
91-10	07/08/91	Explosive Searches at Protected Area Portals	COL Applicant/Holder
91-11	07/19/91	Resolution of Generic Issue 48, "LCOs for Class 1E Tie Breakers", Pursuant to 10 CFR 50.54(f)	Section 1.11
91-14	09/23/91	Emergency Telecommunications	COL Applicant
91-15	09/23/91	Operating Experience Feedback, Solenoid-Operated Valve Problems at U.S. Reactors	COL Holder
91-16	10/03/91	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	COL Applicant
91-17	10/17/91	Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants"	COL Holder Section 1.11
92-01r1		Reactor Vessel Structural Integrity	Subection 5.3.2 and 5.3.3

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
92-04	8/19/92	Resolution of the Issues Related to Reactor Vessel Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)	The ESBWR includes a continuous purge of water to the reference leg to prevent the buildup of non-condensable gases. The CRD Hydraulics provides this flow. Subsections 4.6.1.2.4 and 7.7.1.2
92-08	12/17/92	Thermo-Lag 330-1 Fire Barriers	Not Applicable. The ESBWR provides strict physical separation between the redundant safety-related divisions. There is no need to use Thermo-Lag 330. Subsection 9.5.1
93-05	9/27/93	Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation	Not Applicable. Is an administrative communication. Lessons from the Tech Spec Improvement programs have been factored into the proposed ESBWR Tech Specs. Section 16.
93-06	10/25/93	Research Results on Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas"	The ESBWR only uses highly combustible gases in any safety-related area for reference gas in the H2/O2 monitors. This calibration gas is only used periodically and normally valved out of service. The H2 bottles are located in a non-safety structure. The lines to the H2 monitors are very small and would limit the flow in the event of a break.
93-08	12/29/93	Relocation of Technical Specification Tables Of Instrument Response Time Limits	Not Applicable. Is an administrative communication.
94-01	5/31/94	Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators	The ESBWR does not have safety-related emergency diesel generators. There are no surveillance requirements for the non-safety diesel generators.
94-02	7/11/94	Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in BWRs	The ESBWR addresses the concerns of Thermal-Hydraulic Instability. Section 4.4 and Appendix 4D

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
94-03	7/25/94	Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors	Controls on material properties and welding parameters are placed on all stainless material used in the RPV including the shroud. Subsection 5.2.3.4.1.
95-07	8/17/95	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	The number of safety-related valves in the ESBWR is much smaller than previous designs. The safety-related valves that need to open to perform their function are even smaller. Many of the safety-related valves that need to open are squib actuated and not subject to this phenomenon. Globe valves are generally used in the other applications. In any case, GL 89-10 Supplement 6 now covers this issue and the ESBWR complies with the guidance of this document. Subsection 3.9.6
96-01	1/10/96	Testing of Safety-Related Circuits	The passive systems utilize safety-related electrical busses that are designed as un-interruptible power sources. There is no load shedding and sequencing in the safety-related electrical systems.
96-03	1/31/96	NRC Generic Letter 96-03: Relocation of The Pressure Temperature Limit Curves And Low Temperature Overpressure Protection System Limits	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.0) COL Holder scope. Subsection 16.5.6.4
96-04	6/26/96	Boraflex Degradation in Spent Fuel Pool Storage Racks	The equipment specification for the racks at the time of the order will be consistent with the latest regulatory guidance. Subsection 9.1.2
96-05	9/18/96	Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves	COL Holder to supplement Subsection 3.9.6
96-06	9/30/96	Assurance of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions	Subsections 6.2.1, 6.2.2 and 6.2.4

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
96-06s1	11/13/97	NRC Generic Letter 96-06, Supplement 1: Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	Subsections 6.2.1, 6.2.2 and 6.2.4
97-04	10/7/97	NRC Generic Letter 97-04: Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	Not applicable, the ESBWR does not have ECCS nor safety-related containment cooling pumps
98-01	5/11/98	NRC Generic Letter No. 98-01: Year 2000 Readiness of Computer Systems at Nuclear Power Plants	Outdated concern
98-01s1	1/14/99	NRC Generic Letter No. 98-01 Supplement 1: Year 2000 Readiness of Computer Systems at Nuclear Power Plants	Outdated concern
98-04	7/14/98	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	Not applicable to the ESBWR GDCS. The GDCS pools do not have the debris transport mechanisms that the Suppression Pool is subject to. The PCCS pools are not subject to LOCA debris. There is no safety-related containment spray.
99-02	6/3/99	NRC Generic Letter 99-02: Laboratory Testing of Nuclear-Grade Activated Charcoal	COL Applicant/Holder Subsection 9.4.1
03-01	6/12/03	Control Room Habitability	The verification requirements are in accordance with the applicable regulatory guidance and standards. See Section 6.4
Type: Bulletins			
79-02r2	3/8/79	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	COL Applicant/Holder
79-08	4/14/79	Events Relevant to BWR Identified During TMI Incident	Appendix 1A
80-01	1/11/80	ADS Valve Pneumatic Supply	The design of the pneumatic supply to the ADS valves addresses the concerns with the potential loss of loss of pneumatic pressure. In addition the ESBWR has diverse means of depressurizing the RPV using the DPVs.

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
80-03	2/6/80	Loss of Charcoal from Absorber Cells	Subsection 9.4.1
80-05	3/10/80	Vacuum Condition Resulting in Damage to Chemical and Volume Control System (CVCS) Holdup Tanks	Not applicable to BWRs
80-06	3/13/80	ESF Reset Controls	Section 7.3
80-08	4/7/80	Containment Lines Penetration Welds	COL Applicant/Holder
80-10	5/6/80	Non-Radioactive System – Potential for Unmonitored Release	Subsections 9.2.1 and 9.2.2
80-12	5/9/80	Decay Heat Removal System Operability	Not applicable to BWRs
80-13	5/12/80	Cracking in Core Spray Spargers	Not Applicable. The ESBWR does not have a core spray sparger.
80-15	6/18/80	Possible Loss of Emergency Notification System with Loss of Offsite Power	COL Applicant/Holder
80-20	7/31/80	Westinghouse Type W-2 Switch Failures	Not applicable to new equipment
80-21	11/6/80	Valve Yokes Supplied by Mole	Not applicable to new equipment
80-22	9/11/80	Automatic Industries, Model 200-500-008 Sealed Source Containers	Not applicable to new equipment
80-24	11/21/80	Prevention of Damage due to H2O Leakage Inside Containment	Not applicable, the ESBWR Containment is cooled using the Chilled Water System (HCW) which is closed loop system.
80-25	12/19/80	Operating Problems with Target Rock SRVs at BWRs	Not applicable to the ESBWR design. Different valve type to be used. Subsection 5.4.13
81-01	1/27/81	Surveillance of Mechanical Snubbers	COL Holder

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
81-02	4/9/81	Failure of Gate Type Valves to Close	Not applicable to BWRs
81-02, Supp 1	8/19/81	Failure of Gate Type Valves to Close Against Differential Pressure	Not applicable to BWRs
81-03	4/10/81	Flow Blockage of Cooling Water to Safety System	Not applicable
82-04	12/3/82	Deficiencies in Primary Containment Electrical Penetration Assemblies	Subsection 8.3.4.7
83-06	7/22/83	Non-Conforming Materials Supplied by Tube-Line Corp.	Not applicable, vendor supply issue
84-01	2/3/84	Cracks in Boiling Water Reactor Mark 1 Containment Vent Headers	Not applicable to the ESBWR containment design
84-03	8/24/84	Refueling Cavity Water Seal	The ESBWR will utilize permanently installed flexible bellows between the RPV and the refueling cavity.
85-03	11/15/85	Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings	COL Applicant
85-03, Supp 1	4/27/88	Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings Past Related Correspondence: IE Bulletin 85-03, IE Notice 86-29, and IE Notice 87-01	COL Applicant
86-01	5/23/86	Minimum Flow Logic Problems That Could Disable RHR Pumps	Not Applicable. The ESBWR does not have safety-related RHR pumps
86-03	10/8/86	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	Not Applicable. The ESBWR does not have ECCS pumps
87-01	7/9/87	Thinning of Pipe Walls in Nuclear Power Plants	Subsection 6.6.7

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
87-02	11/6/87	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
87-02, Supp 1	4/22/88	Fastener Testing to Determine Conformance with Applicable Material Specifications Past Related Correspondence: IE Notice 88-17	COL Applicant
87-02, Supp 2	6/10/88	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
88-04	5/5/88	Potential Safety-Related Pump Loss	Not applicable, the ESBWR does not have safety-related pumps
88-07	6/15/88	Power Oscillations in Boiling Water Reactors (BWRs) Past Related Correspondence: IE Notice 88-39	Sections 4.4 and 4B.7
88-07, Supp 1	12/30/88	Power Oscillations in Boiling Water Reactors (BWRs)	Sections 4.4 and 4B.7
90-01	03/09/90	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	Not applicable, the vendor has corrected the problem and new transmitters have been changed to correct the problem.
90-02	03/20/90	Loss of Thermal Margin Caused by Channel Box Bow	Section 4B.2
92-01	6/24/92	Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage	Not Applicable. The ESBWR provides strict physical separation between the redundant safety-related divisions. There is no need to use Thermo-Lag 330.
92-01s1	8/28/92	Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function	Not Applicable. See above.

Table 1C-1

Operating Experience Review Results Summary

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
93-02	5/11/93	Debris Plugging of Emergency Core Cooling Suction Strainers	Not applicable to the ESBWR GDCS. The GDCS pools do not have the debris transport mechanisms that the Suppression Pool is subject to.
93-02s1	2/18/94	Debris Plugging of Emergency Core Cooling Suction Strainers	Not applicable to the ESBWR GDCS. See above.
93-03	5/28/93	Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs	The ESBWR includes a continuous purge of water to the reference leg to prevent the buildup of non-condensable gases. The CRD Hydraulics provides this flow. Subsections 4.6.1.2.6 and 7.7.1.2
94-01	4/14/94	Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1	The FAPCS is designed to prevent the possibility of draining water from the Spent Fuel Storage Pool. COL Holder
95-02	10/17/95	Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode	Not Applicable. The ESBWR does not have a safety-related suppression pool cooling system.
96-02	4/11/96	Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment	COL Holder to supplement Subsection 9.1.5
96-03	5/6/96	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors	Not applicable to the ESBWR GDCS. See response to BL 93-02
96-04	7/5/96	Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks	Related to dry cask storage, which is not part of the ESBWR design. COL Holder will address if applicable.
2005-02	7/18/05	Emergency Preparedness and Response Actions for Security-Based Events	COL Holder will provide response.

APPENDIX 1D REGULATORY TREATMENT OF NON-SAFETY SYSTEMS

Reference 1D-1, Attachment 2 provides the NRC position on the Regulatory Treatment of Non-Safety Systems (RTNSS). Based on the Attachment 2, Section A.I of Reference 1D-1, the RTNSS basis applies broadly to those nonsafety-related SSCs that perform risk-significant functions, and therefore, are candidates for regulatory oversight. Reference 1D-1, Attachment 2, Section A.I applies the following RTNSS criteria to determine these SSC functions:

A. SSC functions relied upon to meet beyond design basis deterministic NRC performance requirements such as 10 CFR 50.62 for anticipated transient without scram (ATWS) mitigation and 10 CFR 50.63 for station blackout (SBO).

B. SSC functions relied upon to resolve long-term safety (beyond 72 hours) and to address seismic events.

C. SSC functions relied upon under power-operating and shutdown condition to meet the NRC's safety goal guidelines of a core damage frequency (CDF) of less than 1.0E-4 each reactor year and large release frequency (LRF) of less than 1.0E-6 each reactor year.

D. SSC functions needed to meet the containment performance goal (SECY-93-087, Issue I.J), including containment bypass (SECY-93-087, Issue II.G), during severe accidents.

E. SSC functions relied upon to prevent significant adverse systems interactions.

1D.1 REVIEW OF RTNSS CRITERIA

Each of the above RTNSS criteria has been reviewed against the ESBWR design. The summary results of those reviews are provided below.

Criterion A

The Standby Liquid Control (SLC) system, used to mitigate an ATWS, is classified as safety-related. As a result, RTNSS Criteria A does not apply to the SLC system. The duration of an SBO for the ESBWR is assumed to be 8 hours, while the ESBWR's capability to mitigate a SBO using only safety-related equipment is 72 hours. Therefore, for the SBO event, no equipment needs to be included within the RTNSS program.

Criterion B

After 72 hours, the only function required for maintaining the plant in a safe shutdown condition is to provide makeup water to the Passive Containment Cooling (PCC), Isolation Condenser (IC) and Spent Fuel pools. Permanently installed piping is included in the Fuel and Auxiliary Pool Cooling System (FAPCS), which is connected directly with the site Fire Protection System (FPS). This connection enables the pools to be filled with water from FPS to continue decay heat removal nearly indefinitely. The FPS has access to enough water on-site to provide makeup water to extend the cooling period from 72 hours through 7 days. Additional sources of readily available water will be identified by the Combined Operating License (COL) licensee. (See Subsection 1D.4.1.)

In addition, FAPCS also includes permanent piping with connections outside the Reactor and Fuel Buildings, which allow alternate water, sources to be used to fill the pools. The portions of FAPCS that connect with the FPS and external building connections and are used to re-fill the PCC, IC and Fuel pools; are classified as safety-related.

The FPS is described in DCD Subsection 9.5.1 and has two diesel and one motor driven pump. The FPS is classified as nonsafety-related but is designed so that portions of the system remain operable following a seismic event. These portions include the diesel driven pump in the Fire Protection Enclosure (FPE), the water supply, the suction pipe from the water supply to the pump, one of the supply pipes from the FPE to the Reactor Building and the connections to the FAPCS. However, Section 1.D.4 has actions for a COL licensee to provide assurance that the site specific design has the necessary reliability and capacity to fill the pools, post 72 hours.

Therefore, RTNSS Criterion B only applies to selected portions of the ESBWR fire protection system.

The Basemat-Internal Melt Arrest Coolability (BiMAC) system is provided in the event of a severe accident to ensure coolability of corium that melts through the bottom of the reactor vessel and relocates to the lower drywell floor. The BiMAC is designed to assure that the primary containment maintains its functionality in the event of a severe accident. Previous designs have demonstrated accepTable containment performance by a defined spreading the core on the area, use of basaltic concrete, and flooding of the lower drywell. The addition of the BiMAC eliminates the uncertainty previously associated with the acceptance of the previous design provisions.

Criterion C

Because of the design margins, redundancy and diversity of the ESBWR systems and components, the probabilistic risk assessment (PRA) summarized in Chapter 19 shows that no nonsafety-related component failure can result in a CDF > 1.0E-4 or a LRF > 1.0E-6 during the initial 72 hours. In the event that multiple failures prevent opening the valves connecting the equipment pool and reactor cavity to both PCC/IC buffer pools, establishing a connection to just one of the two buffer pools is sufficient to replenish the water using FPS and FAPCS and ensure cooling after 24 hours. These criteria continue to be satisfied beyond 24 hours as long as the PCC and IC pools can be filled. Therefore, the Criterion C safety concerns have already been addressed for the generic design of the ESBWR.

Criterion D

A deterministic resolution on meeting the containment performance goal in SECY-93-087, Issue I.J is addressed in detail in Subsection 6.2.5.4 and Chapter 8 of "ESBWR Design Certification Probabilistic Risk Assessment" (Reference 1D-3). A discussion addressing the containment bypass issue from SECY-93-087, Issue II.G, during severe accidents, is located within Subsection 6.2.1.1.5. The Criterion D safety concerns have already been addressed for the ESBWR, and no additional RTNSS related effort is needed.

Criterion E

For the ESBWR, where the failure of a nonsafety-related system or component could cause a safety-related component or system to fail to perform a safety-related function, the nonsafety-related to safety-related interface component is classified as safety-related or has augmented requirements applied, such as requiring a nonsafety-related pipe/component to be seismically qualified or have seismic supports to ensure the protection of a nearby safety-related component. Table 1D-2 provides additional examples on how the safety concern in Criterion E is addressed. Therefore, the Criterion E safety concern has already been resolved for the ESBWR.

1D.2 SPECIFIC STEPS IN THE RTNSS PROCESS

1. Comprehensive Baseline PRA

A comprehensive Level 3 PRA (baseline PRA), in accordance with the EPRI Utility Requirements Document for Advanced Light Water Reactor (ALWR) designs has been prepared for the ESBWR. The results are summarized in DCD Tier 2 Chapter 19. The COL applicant will prepare an update of the PRA to incorporate site-specific design elements and frequencies. (This COL action is covered in Chapter 19). The baseline PRA includes all appropriate internal and external events for both power and shutdown operations. Seismic events are evaluated by a margins approach. Adequate treatment of uncertainties, long-term safety operation, and containment performance are included. Containment performance is addressed with considerations for sensitivities and uncertainties in accident progression and inclusion of severe accident phenomena, including explicit treatment of containment bypass. Mean values are used to determine the availability of passive systems and the frequencies of core damage and large releases. Appropriate uncertainty and sensitivity analyses are used to estimate the magnitude of potential variations in these parameters and to identify significant contributors to these variations.

2. Search for Adverse Systems Interactions

Adverse interactions between the active and passive systems have been systematically evaluated in the process of designing and testing the passive systems. The results of this analysis have been used for design improvements to minimize adverse systems interaction, and as such, are considered in formulating the PRA models. Known adverse system interactions in active non-safety systems from previous GE designs have been addressed in the design of the ESBWR

3. Focused PRA

The focused PRA includes the passive systems and only those active systems necessary to meet the safety goal guidelines proposed by EPRI in scope Criteria I.C. The following is considered in constructing the focused PRA, to determine the reliability/availability (R/A) missions of nonsafety-related SSCs, which are risk significant.

First, the scope of initiating events and their frequencies are maintained in the focused PRA as in the baseline PRA. As a result, nonsafety-related SSCs used to prevent the occurrence of initiating events may be subject to the RTNSS controls.

Second, following an initiating event, the comprehensive Level 3 focused PRA event tree logic does not include the effect of nonsafety-related SSCs. As a minimum, these event trees do not include the defense-in-depth functions and their support such as ac power to determine if the passive safety systems, when challenged, can provide sufficient capability without non-safety backup to meet the NRC safety goal guidelines for a core damage frequency of 1.0E-4 each year and a large release frequency of 1.0E-6 each year. Within Section 6.2, the containment performance evaluations include potential bypass, during accidents. Nonsafety-related systems and components, which remain in the focused PRA model, based on their risk significance are included within Table 1D-1.

4. Selection of Important Nonsafety-Related Systems

Combinations of nonsafety-related SSCs that are necessary to meet NRC regulations, safety goal guidelines, and the containment performance goal objectives have been determined. These combinations are determined for both scope Criteria A and E where NRC regulations are the bases for consideration, and scope Criteria C and D where PRA methods are the bases for consideration. To address the long-term safety issue in scope Criterion B**,** PRA insights, sensitivity studies, and deterministic methods are used to establish the ability of the design to maintain core cooling and containment integrity beyond 72 hours. Non-safety SSC functions required to meet beyond design basis requirements (Criterion A), to resolve the long-term safety and seismic issues (Criterion B), and to prevent significant adverse interactions (Criterion E) are addressed in Table 1D-1.

The following steps are taken in using the focused PRA to determine the nonsafety-related SSCs important to risk:

a. Those nonsafety-related SSCs needed to maintain initiating event frequencies at the comprehensive baseline PRA levels are determined.

b. The necessary success paths with nonsafety-related systems and functions in the "focused PRA" to meet the safety goal guidelines, containment performance goal objectives, and NRC regulations are added. Systems are chosen by considering the factors for optimizing the overall design, and the effect and benefit to the particular systems. PRA importance studies assist in determining the importance of these SSCs.

5. Nonsafety-Related System Reliability/Availability Missions

From the focused PRA the functional R/A missions of active systems needed to meet the safety goal guidelines, containment performance goals, and other NRC performance requirements as described in Step 4 are determined. These systems and components have reliability/availability specifications, based on the importance to safety of their functional R/A missions (see Table 1D-1).

1D.3 CONCLUSION

All of the safety issues with regard to the RTNSS applicability criteria have been resolved for the design certification submittal for the ESBWR, and thus, no additional RNTSS process is needed.

1D.4 COL INFORMATION

The COL applicant shall review its plant against the RNTSS criteria in Table 1D-1. As needed, the COL licensee shall implement a RNTSS program that will provide the required level of reliability of system(s) required to keep the PCC, IC and Fuel Pools filled. This includes identifying other readily accessible and suitable volumes of water.

1D.5 REFERENCES

1D-1 USNRC, "Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems (RTNSS) In Passive Plant Designs (SECY-94-084)," SECY-95-132, May 22, 1995.

1D-2 USNRC "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactors (ALWR) Designs", SECY-93-087, April 2, 1993.

1D-3 GE Nuclear Energy, "ESBWR Design Certification Probabilistic Risk Assessment", NEDC-33201, Class III (Proprietary), Revision 0, September 2005 and NEDO-33201 Class I (non-proprietary), Revision 1, scheduled March 2006.

Table 1D-1

Systems and Components That Qualify For RTNSS

System or Component	Defense-In-Depth Function	Specified Reliability/Availability *
PCC, IC and Fuel Pool fill		
Portions of the Fire Protection Systems (FPS)	Provide make-up water to the PCC and IC pools to maintain passive cooling beyond 24 hours from the initiating event in the event that all of the connecting valves from the Equipment Pool and Refueling Cavity fail to open.	Defense-in depth of a CCF causing loss of passive cooling. Reliability will be determined based on the post 72 hour mission.
Portions of the Fire Protection Systems (FPS)	Provide make-up water to the PCC, IC and Fuel pools to extend passive cooling to at least 7 days from the initiating event.	Not required during the first 72 hours. COL applicant to address for the 72 hour to 7-day mission how reliability and availability of make-up water will be provided.
Core Debris Cooling		
BiMAC	Provide cooling of melted core following a severe accident that has melted through the bottom head and relocated to the Lower Drywell.	See Subsection 19.3.4

* To meet NRC safety goals.

Table 1D-2

Examples of Design Features The Prevent Significant Adverse Systems Interactions

Non-safety system	Prevention of adverse system interaction	Examples
Instrument Air	Safety-related components that utilize instrument air are either designed to fail to a safe state; AND/OR	HCU Scram valves; MSIVs
	Are provided with accumulators to store the required gas volume and pressure to ensure that the safety-related function can be performed.	MSIVs
High Pressure Nitrogen	Safety-related components that utilize high pressure nitrogen are either designed to fail to a safe state; AND/OR	MSIVs
	Are provided with accumulators to store the required gas volume and pressure to ensure that the safety-related function can be performed; OR	ADS SRVs MSIVs
	The system has a diverse means of achieving the required action.	IC condensate return valves